U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.2.8 FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Evaluation Branch (AEB)

I. AREAS OF REVIEW

The transient that results from a postulated feedwater line break is sensitive to the break discharge rate; consequently, a range of break sizes should be evaluated both inside and outside containment to determine the acceptability of the response. Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break) or a reactor coolant system heatup (by reducing feedwater flow to the affected steam generator). Therefore, analyses of various postulated break sizes and locations are needed to identify the particular situation that is most limiting with respect to system effects.

If a feedwater line rupture causes the water in the steam generator to be discharged through the break, the water will not be available for decay heat removal after reactor scram. The break location and size may be such to prevent addition of any feedwater to the affected steam generator. An auxiliary feedwater system is therefore provided to assure that feedwater is available to provide decay heat removal.

The review includes evaluation of the applicant's postulated initial core and reactor conditions pertinent to the feedwater line break, the methods of thermal and hydraulic analysis, the postulated sequence of events including analyses to determine the time of reactor trip and time delays prior and subsequent to initiation of reactor protection system actions, the assumed response of the reactor coolant and auxiliary systems, the functional and operational characteristics of the reactor protection system in terms of its effects on the sequence of events, and all operator actions required to secure and maintain the reactor in a safe shutdown condition. The results of the analyses are reviewed to ensure that the values of pertinent system parameters, discussed in subsection II below, are within expected ranges. The parameters of importance for these transients include reactor coolant system pressure, steam generator pressure, fluid temperatures, fuel and clad temepratures, break discharge flow rate, steamline and feedwater flow rates, safety and relief valve flow rates, pressurizer and steam generator water levels, mass and energy transfer within the containment (for breaks inside containment), reactor power, total core reactivity, hot and average channel heat flux, and minimum departure from nucleate boiling ratio (DNBR).

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and ICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe conditions.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model. RSB also reviews the values of all the parameters used in the analytical model. CPB reviews the initial conditions of the core and all nuclear design parameters. This includes power level, power distribution, Doppler coefficients, moderator temperature coefficients, void coefficients, reactor kinetics, DNB correlations, and control rod worth.

A secondary review is performed by the Accident Evaluation Branch (AEB) and the results are used by ASB to complete the overall evaluation of the break analysis. The AEB evaluates the fission product release assumptions used in determining any offsite releases and verifies that the radiological consequences resulting from a feedwater pipe break are within acceptable limits as part of its primary review responsibility for SRP Section 15.6.5. The result of AEB's analysis is transmitted to RSB for use in the SER writeup.

In addition, the RSB will coordinate other branches' evaluations that interface with the overall review of feedwater system pipe breaks as follows: The Auxiliary Systems Branch (ASB) reviews the auxiliary feedwater system to verify that it can function following a feedwater line break, given a single active component failure and with either onsite or offsite power as part of its primary review responsibility for SRP Section 10.4.9. RSB reviews the auxiliary feedwater system to confirm that the flow provided is acceptable for controlling the transient following a feedwater line break. The Mechanical Engineering Branch (MEB) evaluates potential water-hammer effects on safety valve integrity as part of its primary review responsibility for SRP Section 3.9 series. The Containment Systems Branch (CSB) reviews the methodology which evaluates the response of the containment to breaks of feedwater lines with regard to the effects of pressure and temperature on the containment functional capabilities as part of its primary review responsibility for SRP Section 6.2.1. The ICSB reviewer concentrates on the instrumentation and control aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. ICSB also evaluates potential bypass modes and the possibility of manual control by the operator as part of its primary reactor responsibility for SRP Sections 7.1 through 7.7.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP sections of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

The basic objective of the review of feedwater system pipe break events is to confirm that the reactor primary system is maintained in a safe status for a range of feedwater line breaks up to and including a break equivalent in area to the double-ended rupture of the largest feedwater line.

RSB acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criteria 27 and 28, as they relate to the reactor coolant system being designed with appropriate margin to assure that acceptable fuel design limits are not exceeded, and that the capability to cool the core is maintained.

B. General Design Criterion 31, as it relates to the reactor coolant system being designed with sufficient margin to assure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

C. General Design Criterion 35, as it relates to the reactor cooling system and associated auxiliaries being designed to provide abundant emergency core cooling.

D. 10 CFR Part 100, as it relates to the calculated doses at the site boundary.

In addition, task action plan items that are necessary to meet the requirements to maintain adequate decay heat removal and reactor coolant pump integrity and operation are items II.E.1, II.K.2.1, II.E.1.2, II.K.2.8, II.K.3.5, II.K.2.16, II.K.3.25, and II.K.3.40 of NUREG-0718 and NUREG-0737 (Refs. 6 and 7). Specific criteria necessary to meet the relevant requirements of these regulations are as follows:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 3) for low probability events and below 120% of the design pressures for very low probability events such as double-ended guillotine breaks.

2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. Any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines.

4. The integrity of the reactor coolant pumps should be maintained, such that loss of ac power and containment isolation will not result in seal damage.

5. The auxiliary feedwater system must be safety grade and automatically initiated when required.

6. Tripping of the reactor coolant pumps should be consistent with the resolution to TMI Action Plan Item II.K.3.5.

There are certain assumptions which should be used in the analysis regarding important parameters that describe initial plant conditions and postulated system failures. These are listed below.

a. The power level assumed and number of loops operating at the initiation of the transient should correspond to the operating condition which maximizes the consequences of the accident. These assumed initial conditions will vary with the particular nuclear steam supply system and sensitivity studies will be required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report and referenced if considered applicable.

b. The assumptions as to whether offsite power is lost and the time of loss should be made conservatively. Offsite power may be lost simultaneously with the occurrence of the pipe break, the loss may occur during the accident, or offsite power may not be lost. A study should be made to determine the most conservative assumption appropriate to the plant design being reviewed. The study should take account of the effects that loss of offsite power has on reactor coolant and main feedwater pump trips and on the initiation of auxiliary feedwater, and the resulting modification of the sequence of events.

c. The effects of the postulated feedwater line breaks on other systems (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) should be considered in a manner consistent with the intent of Branch Technical Positions ASB 3-1 and MEB 3-1 (Ref. 5).

d. The worst single active component failure should be assumed to have occurred in the systems required to control the transient.

e. The maximum rod worth should be assumed to be held in the fully withdrawn position, per GDC 25. An appropriate rod reactivity worth versus rod position curve should be assumed.

f. The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

g. The initial core flow assumed for the analysis of the feedwater line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results for the case of a feedwater line rupture inside containment. However, this may not be the most conservative assumption. For example, maximum initial core flow results in increased reactor coolant system cooldown and depressurization, decreased shutdown margin, and an increased possibility that the core will become critical and return to power. Since it is not clear what initial core flow is most conservative, the applicant's assumption should be justified by appropriate sensitivity studies.

h. During the initial 10 minutes of the transient, should credit for operator action be required (i.e., RCP trip), an assessment for the limiting consequence must be performed in order to account for operator delay and/or error.

III. REVIEW PROCEDURES

The procedures below are used during reviews of both construction permit (CP) and operating license (OL) applications. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB and are compared to the initial conditions listed in subsection II of this SRP section. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of reactivity parameters used in the applicant's analysis.

Analytical models should be of sufficient detail to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The equations, sensitivity studies, and models proposed by the applicant are reviewed by RSB.

Credit taken for a reactor trip signal or for actuation of engineered safety features should be reviewed by ICSB to determine the ability of the instrumentation and control systems to respond as assumed under accident conditions.

The ability of the auxiliary feedwater system to supply adequate feedwater flow to the unaffected steam generators during the accident and subsequent shutdown is evaluated by ASB as to availabilty and by RSB as to capability to effect an orderly shutdown. Since auxiliary feedwater system designs are diverse and may require both automatic and manual actuation, preoperational tests should be specified to identify any necessary operator actions and to determine the maximum times permitted for their completion.

To the extent considered necessary, the RSB reviewer evaluates the effect of single active failures of systems and components that may alter the course of the accident. This phase of the review uses the system review procedures described in the standard review plan sections for Chapters 5, 6, 7, 8, and 10 of the SAR. The variations with time during the transient of parameters listed in Sections 15.X.X.3(C) and 15.X.X.4(C) of the Standard Format (Ref. 2) are reviewed. The more important of these parameters for the feedwater line break accident (as listed in subsection I of this SRP section) are compared to those predicted for other similar plants to see that they are within the expected range.

The reviewer confirms that the amount of secondary coolant expelled from the system has been calculated conservatively by evaluating the applicant's methods and assumptions, by comparison with an acceptable analysis performed on another plant of similar design, or by comparison with staff calculations for typical plants which will be available from RSB on request.

The reviewer confirms that a commitment has been made in the SAR to conduct preoperational tests to verify that valve discharge rates and response times including, for example, opening and closing times (delay times) for main feedwater, auxiliary feedwater, turbine and main steam isolation valves, and steam generator and pressurizer relief and safety valves, has been conservatively modeled in the accident analyses. In addition, preoperational testing should include verification of reactor trip delay times, startup delay times for actuation of the auxiliary feedwater system, safety injection signal delay time, and delay times for delivery of any high concentration boron injection required to bring the plant to a safe shutdown condition.

Using the information developed in the review, the AEB reviewer evaluates the radiological consequences of the design basis feedwater line break. This evaluation based on a qualitative comparison with the results of the design basis steam line break, or on a detailed analysis using the approach described in the appendix to SRP Section 15.1.5.

The reliability and operability of the auxiliary feedwater systems (AFWS) are reviewed to assure conformance to the following TMI Action Plan Items (References 6 and 7) as they relate to auxiliary feedwater system performance requirements following feedwater piping failures.

(a) Items II.E.1 and II.K.2.1
(b) Items II.E.1.2 and II.K.2.8

The influence of reactor coolant pump trip during ECCS initiation is reviewed to assure conformance to the TMI Action Item II.K.3.5 (References 6 and 7). Should tripping of the reactor coolant pumps require manual action, delays in operator action must be assessed.

The reliability and integrity of the reactor coolant pump seals during loss of alternating-current power and loss of coolant to the seals (i.e., resulting from containment isolation) are reviewed to assure conformance to the TMI Action Items II.K.2.16, II.K.3.25, and II.K.3.40 (References 6 and 7).

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

> The staff concludes that the consequences of postulated feedwater line breaks meet the requirements set forth in the General Design Criteria 27, 28, 31, and 35 regarding control rod insertability and core coolability, 10 CFR Part 100 guidelines regarding radiological dose at the site boundary, and applicable TMI Action Plan Items. This conclusion is based upon the following:

(a) The applicant has met the requirements of GDC 27 and 28 by demonstrating that the resultant fuel damage was minimal, control rod insertability would be maintained and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio (MDNBR) experienced by any fuel rod was ______, resulting in ___% of the rods experiencing clad performation.

(b) The applicant has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(c) The applicant has met the requirements of GDC 35 with respect to demonstrating the adequacy of the emergency cooling systems to provide abundant core cooling and reactivity control (via boron injection).

(d) The analyses and effects of feedwater line break accidents inside and outside containment, during various modes of operation and with and without offsite power, have been reviewed and evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.

(e) The parameters used as input to this model were reviewed and found to be suitably conservative.

(f) The radioactivity release has been evaluated using the computer code SARA and a conservative description of the plant response to the accident. A decontamination factor of ______ between the water and steam phases and a X/Q value of ________ sec/m^3 has been used in our evaluation of radiological consequences. The calculated doses are presented in Table _____. Technical specification limits on primary and secondary coolant activities will limit potential doses to small fraction of the 10 CFR Part 100 exposure guidelines. The potential doses are within 10 CFR Part 100 exposure guidelines even if the accident should occur coincident with an iodine spike.

(g) The applicant met the requirements of TMI Action Plan Items II.E.1, II.K.2.1, II.E.1.2, and II.K.2.8 with respect to demonstrating the adequacy of the auxiliary feedwater design to remove decay heat following feedwater piping failures.

(h) The applicant met the requirements of TMI Action Plan Items II.K.2.16, II.K.3.25 and II.K.3.40 with respect to demonstrating the integrity and operation of the reactor coolant pumps to withstand the postulated accident.

(i) The applicant met the requirements of TMI Action Plan Item II.K.3.5 with respect to the operation and tripping of the reactor coolant pumps. The assumptions used are conservative and consistent with the generic resolution to Item II.K.3.5.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGs.

VI. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

3. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

4. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

   a. General Design Criterion 27, "Combined Reactivity Control System Capability."

   b. General Design Criterion 28, "Reactivity Limits."

   c. General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

   d. General Design Criterion 35, "Emergency Core Cooling."

5. Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP Section 3.6.1; and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP Section 3.6.2.

6. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses."

7. NUREG-0736, "Clarification of TMI Action Plan Requirements."